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3-step program toward a Reactor Development Facility

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PPPL Colloquium,

April 11, 2007, PPPL, Princeton NJ

[1]This work is supported by US DoE contract No. DE–AC020–76–CHO–3073.

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Abstract

The presently adopted plasma physics concept of magnetic fusion has originated from the idea of providing low plasma edge temperature as a condition for plasma-material interaction. During 30-years of its existence this concept has shown to be not only incapable of addressing practical reactor development needs, but also to be in conflict with fundamental aspects of stationary and stable plasma.

Meanwhile, a demonstration of exceptional pumping capabilities of lithium surfaces on T-11M (1998), discovery of the quiescent H-mode regime on D-IIID (2000), and a 4 fold enhancement of the energy confinement time in CDX-U tokamak with lithium (2005), contributed to a new vision of fusion relying on high edge plasma temperature. The new concept, called LiWalls, provides a scientific basis for developing magnetic fusion.

The talk outlines 3 basic steps toward the Reactor Development Facility (RDF) with DT fusion power of 0.3-0.5 GW and a plasma volume $\simeq 30$ m^3. Such an RDF can accomplish three reactor objectives of magnetic fusion, i.e.,

1. *high power density $\simeq 10$ MW/m^3 plasma regime,*
2. *self-sufficient tritium cycle,*
3. *neutron fluence $\simeq 10 - 15$ MW·year/m^2,*

all necessary for development of the DT power reactor. Within the same mission a better assessment of DD fuel for fusion reactors will also be possible.

The suggested program includes 3 spherical tokamaks. Two of them, ST1, ST2, are DD-machines, while the third one, ST3, represents the RDF itself with a DT plasma and neutron production.

All three devices rely on a NBI maintained plasma regime with absorbing wall boundary conditions provided by the Li based plasma facing components. The goal is to utilize the possibility of high edge temperature plasma with the super-critical ignition (SGI) regime, when the energy confinement significantly exceeds the level necessary for ignition by α-particles. In this regard all three represent Ignited Spherical Tokamaks, suggested in 2002.

Abstract

Specifically, the mission of ST1, with a size slightly larger than NSTX in PPPL but with a four times larger toroidal field, is to achieve the absorbing wall regime with confinement close to neo-classical. In particular, the milestone is $Q_{DT-equiv} \simeq 5$ corresponding to the conventional ignition criterion.

The mission of ST2, which is a full scale DD-prototype of the RDF, is the development of a stationary super-critical regime with $Q_{DT-equiv} \simeq 40 - 50$.

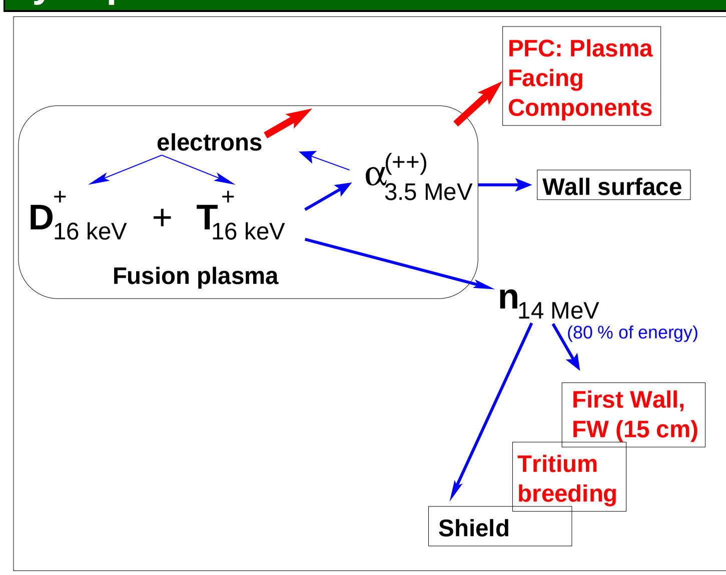
ST3 is a DT device with $Q_{DT} \simeq 40-50$ with sufficient neutron production to design the nuclear components of a power reactor. Still the mission of ST3 contains a significant plasma physics component of developing α-particle power and He ash extraction.

As a motivational step (ST0), the suggested program, assumes a conversion of the existing NSTX device into a spherical tokamak with lithium plasma facing components. The demonstration of complete depletion of the plasma discharge by lithium surface pumping, first shown on T-11M, is considered as a well-defined milestone for readiness of the machine for the new plasma regime. The final mission of ST0 would be doubling or tripling the energy confinement time with respect to the current NSTX.

After 40 years since acceptance of tokamaks as the mainstream approach for magnetic fusion it is the time to introduce significant corrections to our reactor concept.

The old one does not lead to a practical reactor

Mainstream Magnetic Fusion (MMF) relies on plasma heating by α-particles



Flow pattern of fusion energy (since the 50s)

Ignition criterion:

$$f_{pk} \cdot \langle p \rangle \quad \cdot \tau_E^* = 1$$
[MPa · sec]

Peaking factor f_{pk}:

$$f_{pk} \equiv \frac{\langle 16 p_D p_T \rangle}{\langle p \rangle^2}$$

Plasma pressure p:

$$p = p_e + p_D + p_T + p_\alpha + p_I$$

Clean on first glance, the concept has encountered fundamental problems and never approached the nuclear issues of a reactor

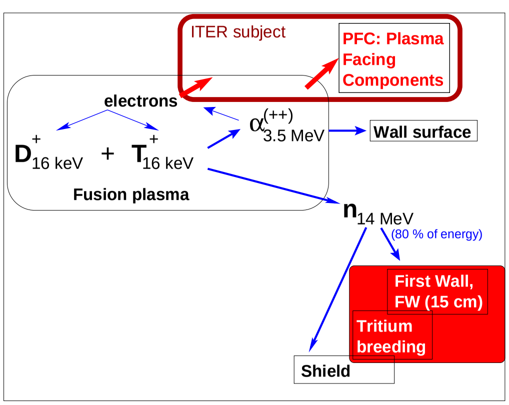
Two fundamental plasma physics problems are hidden in the α-particle heating concept of MMF reactor

1. **Energy from α-particles (20 % of total fusion) is consumed inside the core. Then it is transported to the plasma boundary and released to material surfaces (PFC) in a localized form. No materials exist to withstand the flux $\simeq$ 20 MW/m^2 (without shutting down the plasma).**
2. **90 % of α-particle heat goes to electrons, which do not participate in fusion energy production. At the same time their heat transport is the major channel for energy loss from the plasma.**

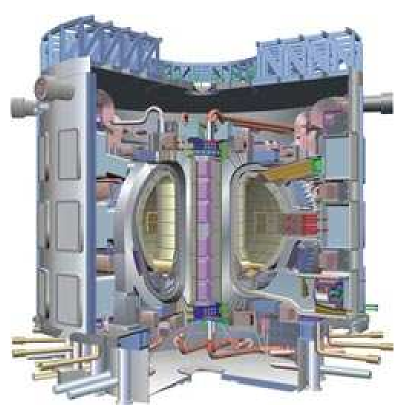
In reactor projections, MMF relies on the "hot-electron" regime, the worst possible one.

All present day machines work in the "hot-ion" mode

ITER is the first machine targeting the α-heating regime



$\Longrightarrow$



Even in expected "burning plasma" regime ITER is still dealing mostly with plasma physics issues.

Being an implementation of the old concept, ITER only barely touches the reactor aspects of fusion

The criterion of conceptual relevance to reactor R&D is very simple: ability of delivering

15 MWa/m^2

of neutron fluence,

or burn-up of

1 kg(T)/m^2 (FW)

(ITER is capable of only 0.3-0.4 MWa/m^2 (burn-up of 10-15 kg of T, instead of 650 kg)

The MMF concept is still in disconnect with R&D for a reactor

Unlike plasma physicists, society recognizes fusion only in conjunction with a reactor

Dana Rohrabacher's shocking notes on funding the fundamental science:

" 1. A focus on long-term, non-commercial R&D with a potential for significant scientific discovery, leaving commercialization to the marketplace.
2. Federal funding should end when technical feasibility is demonstrated. Production should be left to the marketplace.
3. We should fund projects that reflect revolutionary ideas that, if proven, would make possible the impossible within performance-based guidelines.
4. Evolutionary or incremental advances in technology should be handled by the private sector.
5. Each government-owned laboratory should confine in-house research to certain areas of expertise. Other research should be contracted out to industry, foundations and universities.
6. All R&D programs should be tightly focused on the agency's mission. All others should be terminated.

However, we also had to take a serious look at some programs that technically meet these criteria but do not stand up to a cost/benefit test. The Department of Energy's fusion research program is one example.

Over the past 40 years, U.S. taxpayers have paid more than $9 billion for research on fusion energy, yet none of the research has achieved "break-even," the point at which the fusion reaction generates the same amount of energy as is put in. To provide commercial power, a fusion reactor has to generate more energy than is put in, and no scientist has been able to tell me that we will reach that goal in less than 40 more years."

(APS News v.4. no.7, July 1995)

(In the 70s, the Congressman was one of enthusiasts of fusion)

It is certainly possible to confront a Congressman with our scientific achievements, but this does not resolve the issue

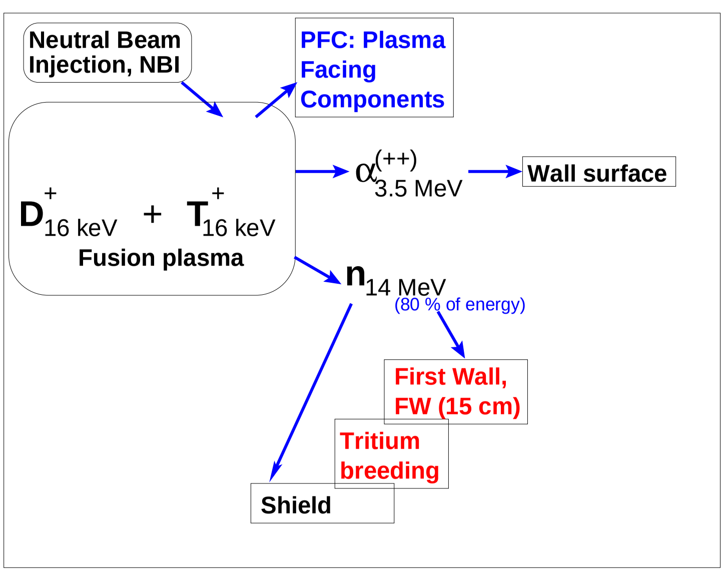
There is a lot of people, who cannot be fooled (by our development plans), and who asks a simple question

When you will give us your energy ?

(My friends Fima and Gustav, April 7, 2007)

Ignoring Congressmen, we all still have to feel the pressure from our friends to deliver highly needed energy

The LiWall Fusion (LiWF) relies on NBI and Li pumping walls



LiWall concept has a clean pattern of flow of fusion energy

α-particles are free to go out of plasma

NBI controls both the temperature and the density

$$P_{NBI} = \frac{3}{2}\frac{\langle p\rangle V_{pl}}{\tau_E},$$
$$\frac{dN_{NBI}}{dt} = \Gamma^{ions}_{core\rightarrow\ edge}$$

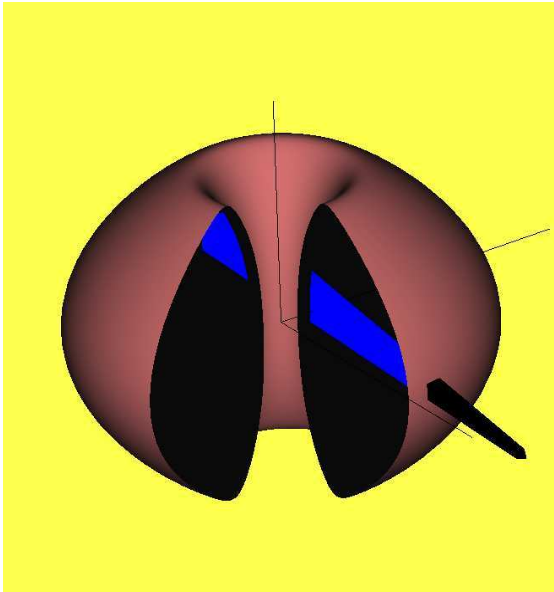
Super-Critical Ignition (SCI) confinement is necessary to make NBI work this way

$$\tau_E >> \tau^*_E$$

Plasma physics issues, unhandleable by MMF, disappear in LiWF

LiWF is suitable for reactor design issues

Neutral Beam Injection (NBI) is a ready-to-go fueling method for pumping wall based magnetic fusion



The energy should be consistent with the plasma temperature

$$E_{NBI} = \left(\frac{3}{2} + 1\right)(T_i + T_e),$$

e.g., for

$$T_e \simeq T_i \simeq 16\ keV$$

$$E_{NBI} = 80\ keV$$

In absence of cold particles from the walls, after collisional relaxation

$$\nu_i = 68\frac{n_{20}}{T_{i,10}^{3/2}}, \quad \nu_e = 5800\frac{n_{20}}{T_{e,10}^{3/2}}$$

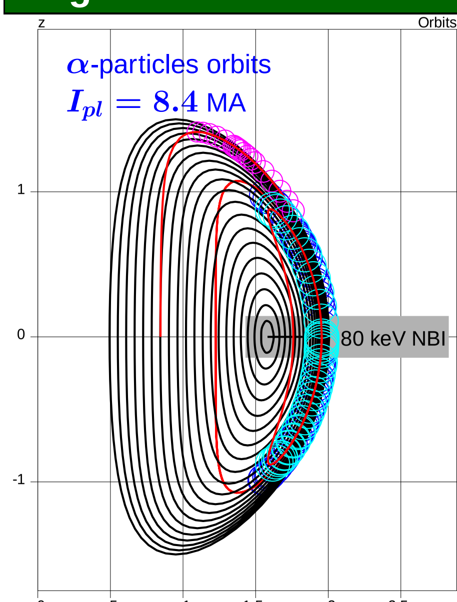
the temperature profile becomes flat automatically

$$T_i = const, \quad T_e = const, \quad T_e < T_i$$

LiWF makes the "hot-ion" mode to be perfect for fusion

A super-critical ignition regime is expected

Large Shafranov shift makes core fueling possible in IST



The charge-exchange penetration length

$$\lambda_{cx} \simeq \frac{0.6}{n_{e,20}} \frac{V_b}{V_{b,80\ keV}}\ [m]$$

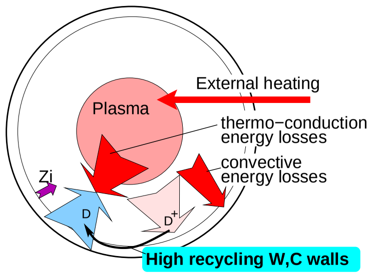
The distance between magnetic axis and the plasma surface in IST

$$R_e - R_0 = 0.3 - 0.5\ [m]$$

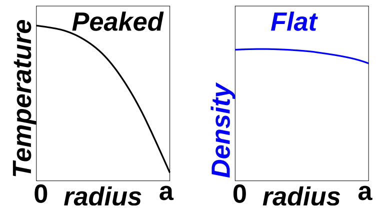
80 keV NBI can provide core fueling and control of fusion power

Even at 8.4 MA 60 % of alphas intersect the plasma boundary and can be intercepted at first orbits

The right plasma-wall contact is the key to magnetic fusion

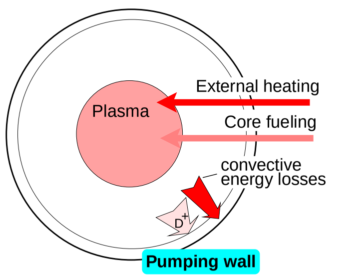


MMF requires a low temperature plasma edge

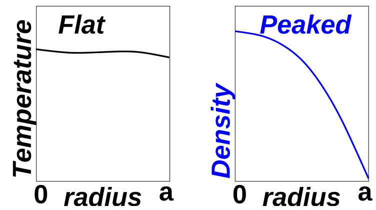


As a "gift" from plasma physics MMF gets ITG/ETG turbulent transport.

Most of the plasma volume does not produce fusion



Molten Li pumps the plasma out. High edge T is OK



No "gifts" from plasma physics (ITG/ETG, sawteeth, ELMs) are expected or accepted.

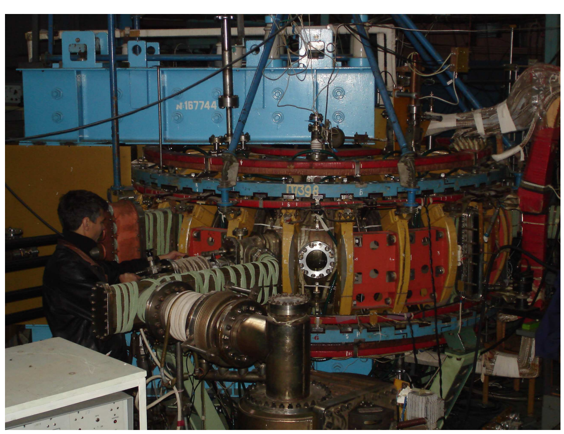
LiWF relies only on external control.

The entire plasma volume produces fusion

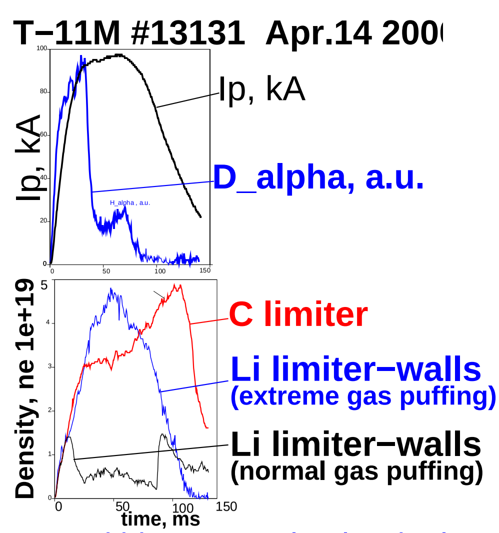
The pumping walls would dramatically simplify plasma-wall interactions and may have an unprecedented impact on the plasma core

In 1998 T-11M tokamak (TRINITI, Troitsk, RF) demonstrated outstanding plasma pumping by Li coated walls

(http://w3.pppl.gov/~zakharov/Mirnov010221/Mirnov.ppt, *p.18, Exper. Seminar PPPL, Feb. 21, 2001)*



T11M and DoE's APEX/ALPS technology programs triggered the idea of LiWalls



Lithium completely depleted the discharge in T-11M

In PPPL, CDX-U demonstrated similar pumping capabilities

Inventory of lithium for pumping purposes is not the issue

E.g., for the ITER size plasma 3-4 L of lithium (0.1 mm $\times$ 30-40 m^2) with the rate of replenishment

$$10L/hour, \quad V_{Li} < 1 \text{ [cm/sec]}$$

is sufficient.

Existing technology of capillary systems (“Red Star”, T-11M, FTU, UCSD), gravity and Marangoni effect provide a solid design basis for pumping surfaces. Everybody has his own experience with solder and copper wire.

The issue is only in the oxidation (hydrolyzation) of the Li surface during the idle period of the machine.

In LiWF molten lithium can be used to control the inventory of unburned tritium

In MMF approach, the gas puffing (in addition to 100% recycling) spreads tritium over all channels inside the machine.

Plasma edge temperature is determined by the particle fluxes rather than by near edge transport properties

S. Krasheninnikov’s boundary conditions

$$\frac{5}{2}\Gamma_e T_e^{edge} = \int_V P_e dV, \qquad \frac{5}{2}\Gamma_i T_i^{edge} = \int_V P_i dV, \qquad T_{i,e}^{edge} \simeq T_{i,e}(0)$$

lead to elimination of the thermo-conduction in energy transport

$$\underbrace{\frac{5}{2}\oint \Gamma_{i,e} T^{i,e} dS}_{convection} + \underbrace{\oint q_{i,e} dS}_{thermo-conduction} = \underbrace{\int_0^V P_{i,e}(V) dV}_{Power\ source}, \qquad \underbrace{\oint q_{i,e} dS}_{thermo-conduction} \simeq 0$$

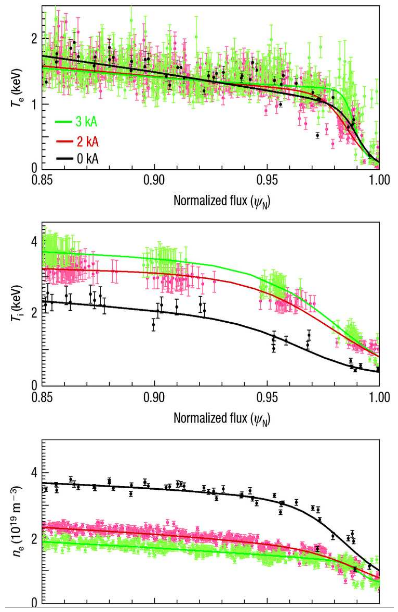
$$\underbrace{\oint \Gamma_{i,e} dS}_{convection} = \underbrace{\int_v S_{i,e} dV}_{particle\ source}$$

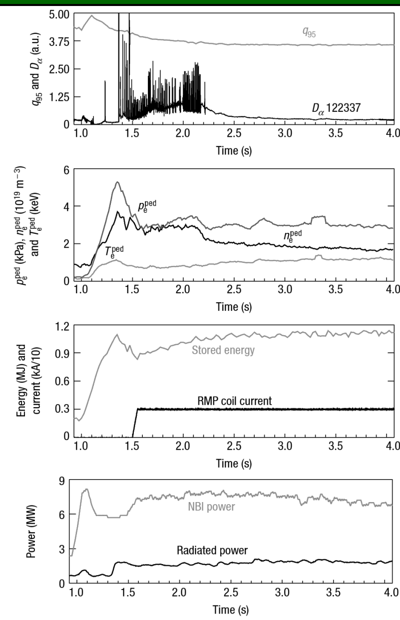
The energy losses from the plasma are exclusively convective and, thus, determined by the best confined component (ions).

The LiWF introduces in fusion the best possible confinement regime

In MMF the energy losses are due to turbulent thermo-conduction

RMP experiments on D-IIID have confirmed our, LiWF, views





0 kA, 2 kA, 3 kA $I_{RMP-coil}$ *T.Evans at al., Nature physics 2, p.419, (2006)*

These observations have dismissed one of the misconceptions of MMF about "edge transport barrier"

LiWF is the only fusion concept not sensitive to the electron thermo-conduction in the plasma core.

Energy losses are determined by particle diffusion

Relative sizes of CDX-U (which quadrupled τ_E with lithium in 2005), NSTX (the holder of the record $\beta = 40$ %, 2004), ITER (with the α-heating dominated regime), and IST (0.2-0.5 GW)

With high β in Spherical Tokamaks a high power density can be achieved.

LiWF does not require the non-existing "high-tech".

LiWF is compatible with existing fusion and general technology and requirements for Reactor Development Facility

A valuable reference transport model is possible for LiWF

Heat flux:

$$q_i = \chi_i^{neo}\nabla T_i$$ *neo-classical ions, plays no role,*

$$q_e = \chi_i^{neo}\nabla T_e$$ *"anomalous" electrons, plays no role,*

Particle flux is controlled by neo-classical ions:

$$\Gamma_{i,e} = \chi_i^{neo}\nabla n$$ *(Ware pinch neglected being a good effect)*

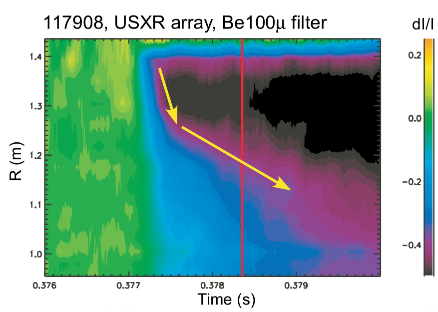
The LiWF does not assume anything regarding confinement of electrons.
It utilizes good confinement of ions, known for decades.

MMF relies exclusively on the "science" of scalings. At the same time, it has no representative database for its "hot-electron" mode

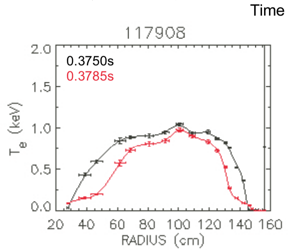
JOHNS HOPKINS UNIVERSITY

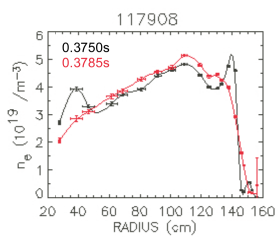
Perturbation Analysis Indicates Two Regions of $\chi_{e,pert}$

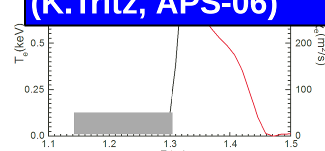
NSTX



- T_e crash propagates from edge to core, n_e globally unperturbed
- Difference in propagation speed corresponds to differences in pertur



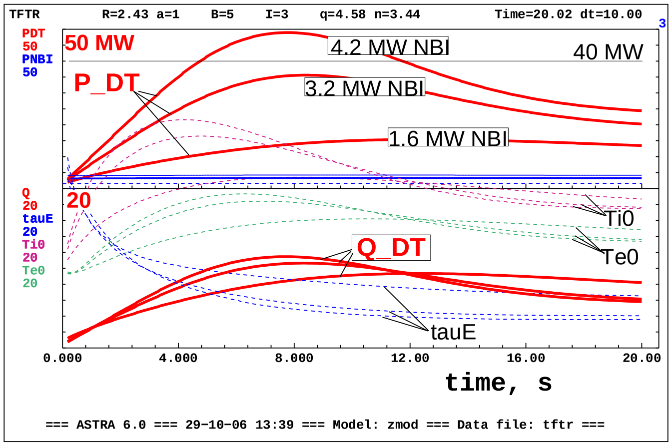




NSTX experiments:
Ions are neo-classical,
Electron are anomalous,
Density profile is not "stiff"
(K.Tritz, APS-06)

- Dependence of $\chi_{e,pert}$ on T_e gradient suggests critical gradient threshold

ASTRA-ESC simulations of TFTR, B=5 T, I=3 MA, 80 keV NBI



Even with no α-particle heating:

$$P_{NBI} < 5 \ [\mathrm{MW}],$$
$$\tau_E = 4.9 - 6.5 \ [\mathrm{sec}],$$
$$P_{DT} = 10 - 48 \ [\mathrm{MW}],$$
$$Q_{DT} = 9 - 12$$

within TFTR stability limits, and with small PFC load (< 5 MW)

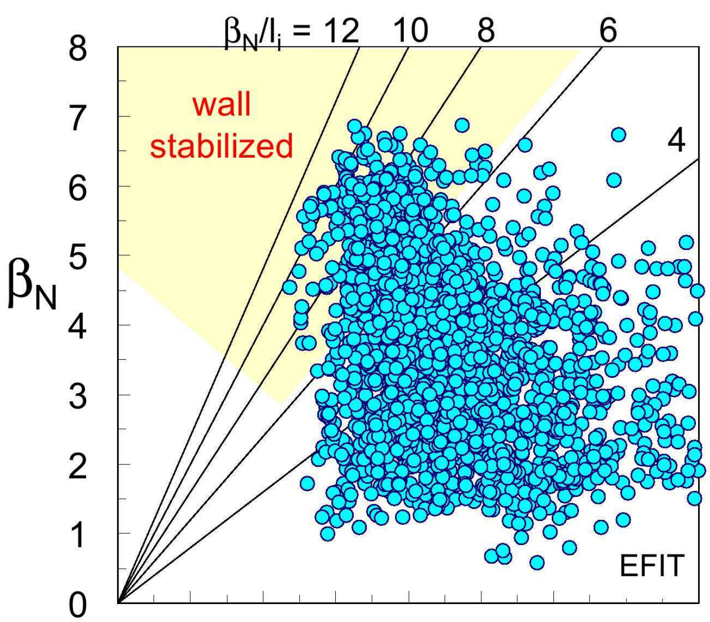
	PNBI	n	T	P DT	Q DT	tauE	nend	Ti0	Te0	gb %
(a)	1.65	0.3	10	15.4	9.34	6.54	0.42	18.7	14.8	1.64
(c)	3.30	0.3	10	35.5	10.6	4.04	0.55	17.6	13.6	1.96
(d)	4.16	0.3	10	48.9	11.6	3.58	0.59	17.5	13.4	1.96

The "brute force" approach ($P_{NBI} = 40$ MW) did not work on TFTR for getting $Q_{DT} = 1$. With $P_{DT} = 10.5$ MW only $Q_{DT} = 0.25$ was achieved.

In the LiWall regime, using less power, TFTR could challenge even the $Q = 10$ goal of ITER

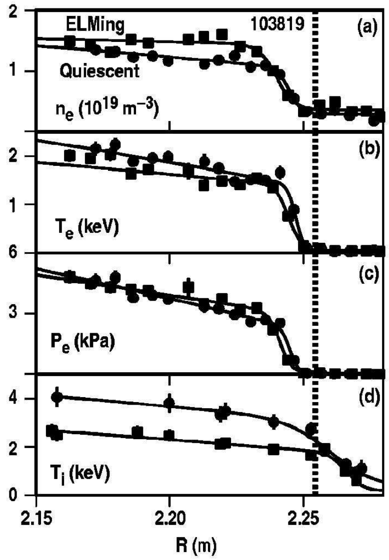
(Ignition criterion corresponds to $Q = 5$)

The stability data base for RDF is already in a good shape

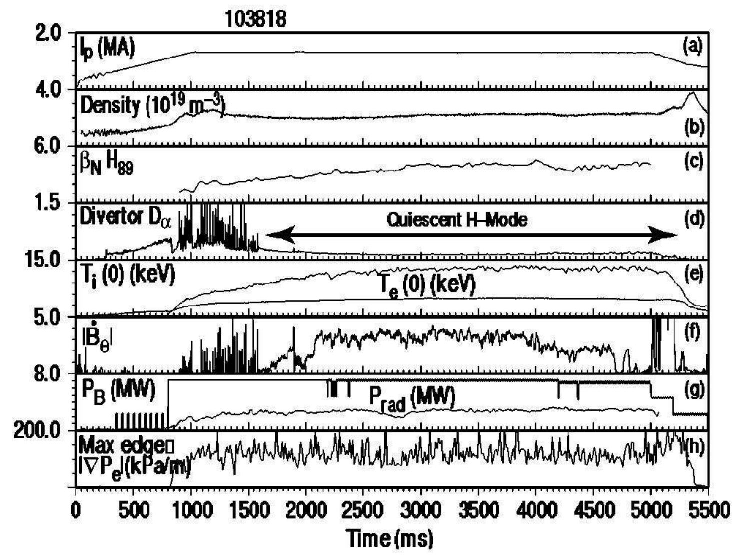


In 2004, beta in NSTX has approached the record level of 40 %

Discovery of the quiescent H-mode in 1999 on DIII-D was a shock (un-noticed by many experts) for MHD theory



Plasma profiles near the edge

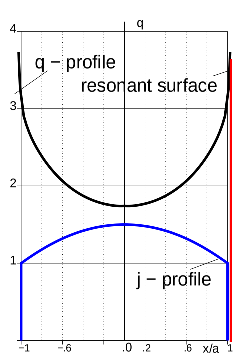


From K.Burrell at al., Phys of Plasmas 8, p.2153, (2001)

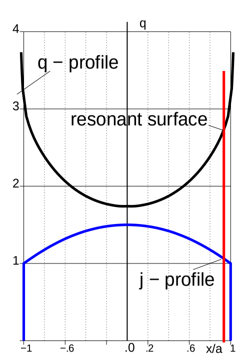
It gave to LiWF extra optimism that free-boundary stability is possible

A widespread belief in MHD theory is that the high edge current density is destabilizing ("peeling modes")

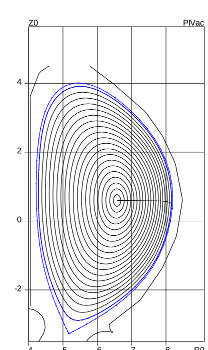
$$W \propto \int \frac{j' R d\rho}{B_{tor}\left(\frac{1}{q} - \frac{n}{m}\right)} \simeq \frac{j_{edge}}{B_{tor}\left(\frac{1}{q_{edge}} - \frac{n}{m}\right)}$$



case 1: $mq_a < n$
Ideally unstable



case 2: $mq_a > n$
Tearing stable

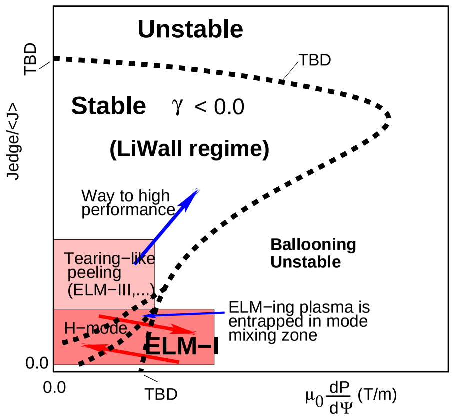


LiWall + Separatrix: $q_a = \infty$
Ideally & tearing stable

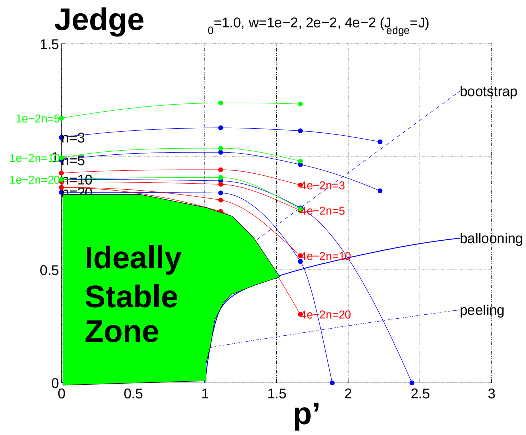
In presence of a separatrix, the finite edge current density is stabilizing

Basic understanding of stability (rather than SciDAC codes) resulted in stability diagram for ELMs

In a wide range, the finite current density at separatrix is stabilizing for ELMs. Pressure is destabilizing.



"Heuristic diagram" (Zakharov, 2005)

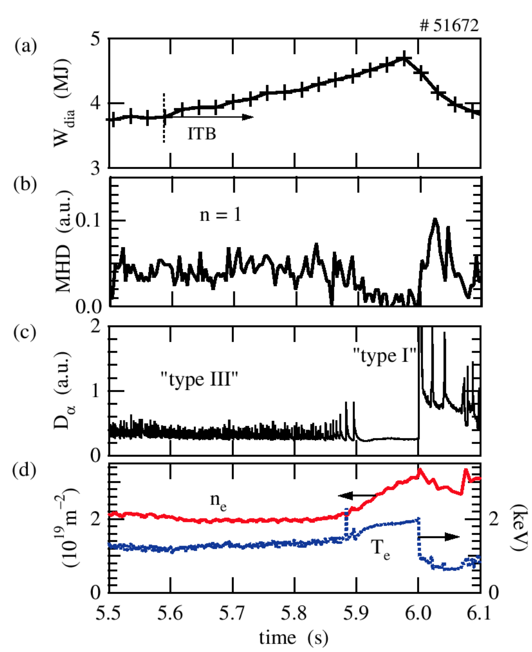


Keldysh Institute calculation, (Medvedev, 2003)

High temperature of LiWF is consistent with the high performance spot on stability diagram

MMF is pushing the operational point right into the mess of ELMs

Quiescent period in JET ITB experiments is consistent with this theory

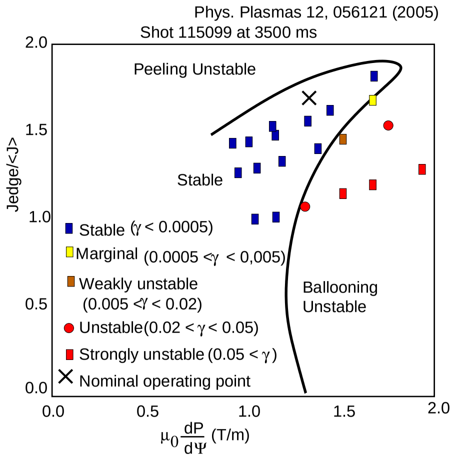


JET has a quiescent regime as transient phase from ELM-III to ELM-I

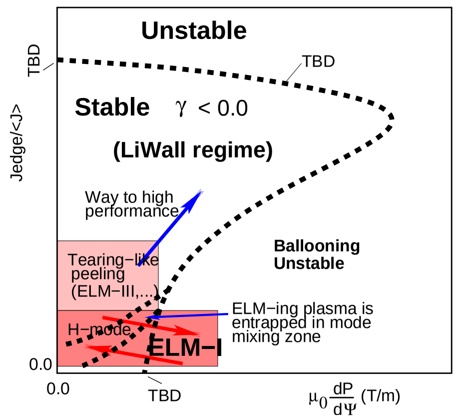
"Edge issues in ITB plasmas in JET"
Plasma Phys. Control. Fusion 44 (2002) 2445-2469 Y. Sarazin, M. Becoulet, P. Beyer, X. Garbet, Ph. Ghendrih, T. C. Hender, E. Joffrin, X. Litaudon, P. J. Lomas, G. F. Matthews, V. Parail, G. Saibene and R. Sartori.

The authors emphasized the crucial role of the edge current density

There are no "peeling" modes for the separatrix limited plasma and $j_{edge} \neq 0$



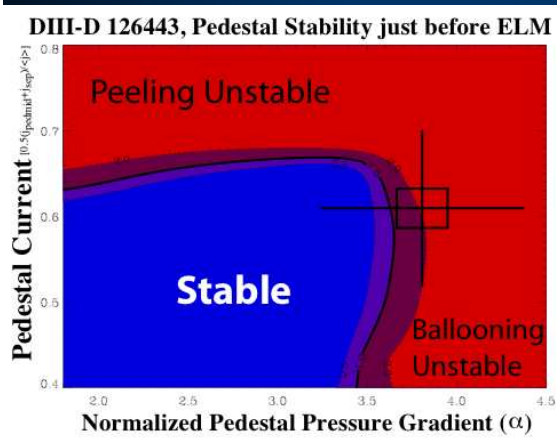
DIII-D interpretation

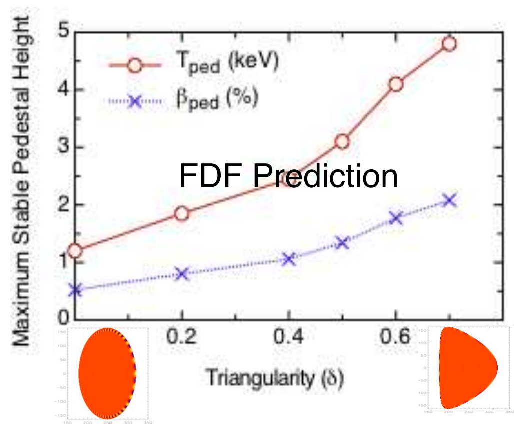


"Zakharov's" diagram (TBD)

GA essentially is switching to our (together with Keldysh Inst.) predictions of ELMs stabilization

FDF Designed with Strong Shaping to Optimize Pedestal Height and Performance





- **Pedestal stability boundaries quantitatively calculated with ELITE code, successfully validated against DIII-D experiments [left]**
  - Maximum pedestal height constrained by peeling-ballooning instabilities which trigger ELMs
- **Pedestal is strongly stabilized by shaping, particularly high elongation and triangularity**
  - Strong shaping can expand apex of stability region to higher pressure
- **FDF design optimized with very high elongation (2.3) and triangularity (0.6)**
  - Leads to more than a factor of three improvement in pedestal height over weak shape
  - Pedestal width remains an uncertainty (typical value 5% of poloidal flux)
  - Control of edge collisionality allows operation near apex of stability region

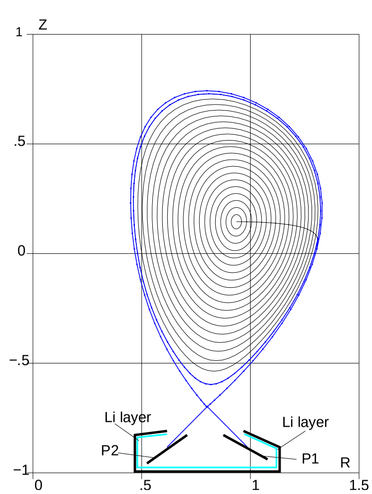
In LiWF there is no tendency of peaking of current density

While, in hunt for discoveries, SciDAC people were calibrating their comprehensive numerical codes against observations of sawteeth on CDX-U (PPPL),

CDX-U discharges became MHD-quiescent after introduction of lithium

Together with the $q = 1$ surface, the LiWall regime wipes out the very opportunity for sawteeth, internal reconnection events, and potentially, for neo-classical tearing modes

LiWF regimes eliminates the effects driving impurities to the plasma core



Three forces are acting on impurities on the way from PFC to the plasma

1. *A small electro-static force ZeE_{SOL}, directed back to the plate.*
2. *Friction $R_V \propto Z^2$ with the ion flow, also directed back to the plate.*
3. *Thermo-force $R_T \propto Z^2$, driving impurities into the plasma.*

In addtion, there is a direct plasma-wall interaction through the radial bursts of blobs

In collisionless SOL the thermo-force is absent, leading to $Z_{eff} \simeq 1$

Blobs are also not expected. There is no indications of blobs in QHM regime on D-IIID

Burn-up of tritium is proportional to the energy confinement time, and can be very efficient in LiWF

$$n \left\langle \sigma v \right\rangle_{DT,16keV} \bar{\tau}_E = 0.03 n_{20} \bar{\tau}_E$$

In LiWF the burn-up of tritium could be a significant fraction of unity

On the other hand

By ingition criterion MMF is locked into very low, 2-3 %, rate of tritium burn-up

Stationary plasma regime integrates the non-inductive current drive, macroscopic stability, density and temperature control, stationary walls and plasma-wall interactions

Being similar to MMF in terms of non-inductive current drive,

LiWF eliminates the root effects affecting the steady state of plasma

For years,

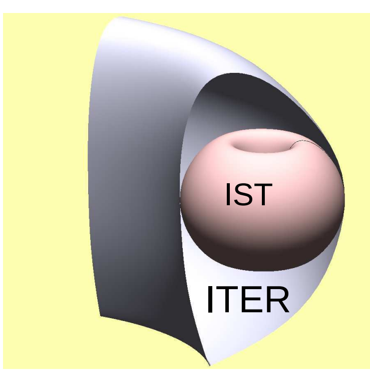
MMF is busy with patching the countless loopholes in its “concept” of the stationary plasma

Unlike MMF, LiWF is consistent with existing plasma physics and technology and has a clean strategy for reactor R&D

Issue	MMF	LiWF
Use of plasma volume	25-30 %	100 %
Fusion producing β_{DT}	$\beta_{DT} < 0.5\beta$	$\beta_{DT} > 0.5\beta$
Anomalous electrons	YES	NO
Transport database	not scalable	scalable
Sawteeth (IREs)	unpredictable	absent
ELMs	unpredictable	absent
Fueling	unresolvable	existing NBI technology
Fusion power control	unpredictable	existing NBI technology
Edge pressure control	reduced performance	RMF, NBI technology
Power extraction	unresolvable	conventional technology
Tritium control	tritium in all channels	pumping by Li
Cost	$\simeq$$20B with no RDF strategy	$2-2.5 B for RDF program

As a reactor concept, the current Mainstream fusion approach is full of fundamental problems, which are stagnating the progress in fusion

The mission of 3-step RDF program is a powerful neutron source for reactor development



RDF should target three mutually linked objectives of magnetic fusion

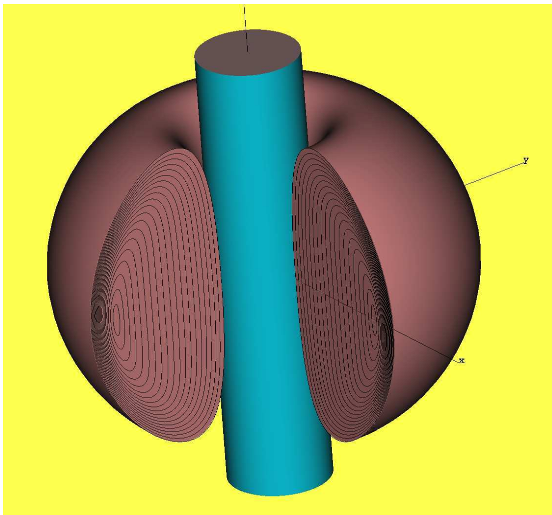
1. *High power density plasma regime regime, $\simeq$ 10 MW/m^3*
2. *Fluence of neutrons 15 MWa/m^2 for designing the First Wall*
3. *Self-sufficient Tritium Cycle*

LiWF approach, together with essentially existing technology, seems to be capable of accomplishing this mission

In the other hand,

“DEMO” with no RDF in the first place is another myth of MMF

Ignited Spherical Tokamaks (IST) are the only candidate for RDF



1. Volume $\simeq$30 m^3.
2. DT power $\simeq$ 0.2-0.5 GW.
3. Neutron coverage fraction of the central pole is only 10 %.
4. FW surface area 50-60 m^2

ITER-like device ($\simeq$ 700 m^2 surface) would have to process 700 kg of tritium for developing the First Wall.

The possibility of an unshielded copper central stack is a decisive factor in favor of IST

Three steps of RDF program ($2-2.5 B) include two DD STs and a final DT machine (not in the Princeton area)

1. ST-1, targeting achievement of the super-critical regime with neo-classical confinement in a DD plasma and

$$Q_{DT}^{equiv} > 5, \qquad f_{pk} \langle p \rangle \tau_E > 1$$

2. ST-2, a full scale DD-prototype of IST for demonstration of all aspects of a stationary super-critical regime with

$$Q_{DT}^{equiv} \simeq 40 - 50$$

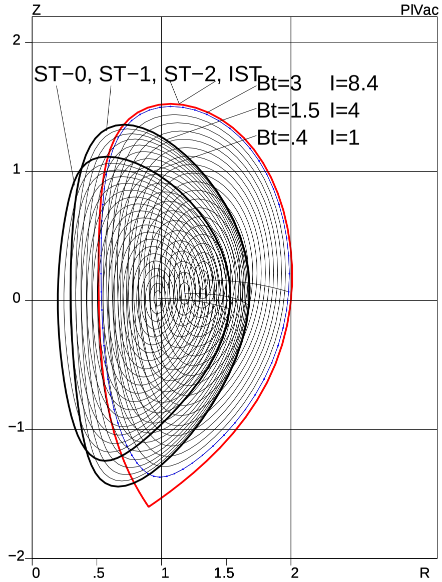
3. ST-3, RDF itself with a DT plasma as a neutron source for reactor R&D and α-particle power extraction studies with

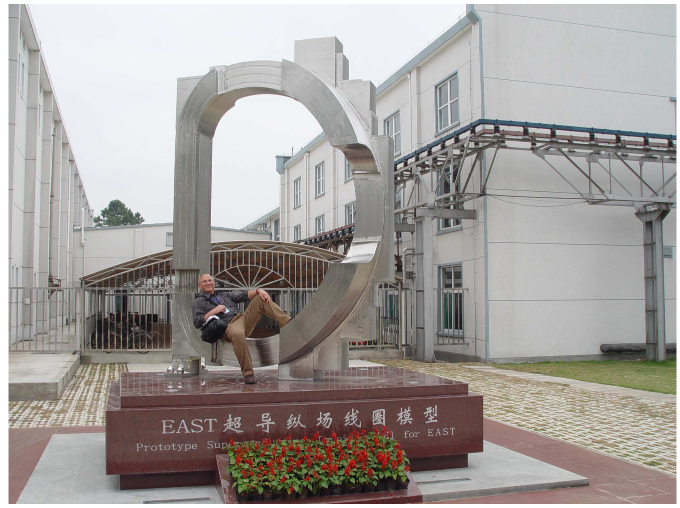
$$Q_{DT} \simeq 40 - 50$$

15 years is a reasonable time for launching ST-3 and to put it in tandem with ITER in order to make the approach to a fusion reactor comprehensive.

Together with ITER RDF can prepare a smooth transtion to the power production

Increase in performance of ST-* is provided by the increase in magnetic field





Superconducting coils are not excluded (at least, for DD STs)

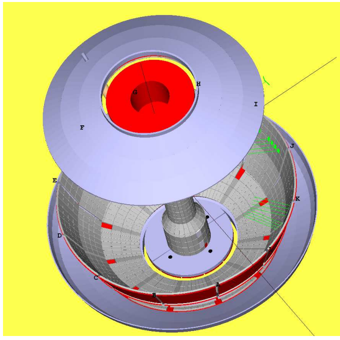
The RDF program assumes conversion of NSTX in PPPL into ST-0 with Li based PFC

- *The current NSTX program is essentially exhausted.*
- *It is focused mainly on self-improvements and is trailing the achievements of other teams, rather than advancing fusion energy.*
- *The program already has been twice explicitly warned about possible shutdown.*
- **On the other hand, the experience accumulated on NSTX, and the machine itself, are extremely valuable for developing the next steps in magnetic fusion.**

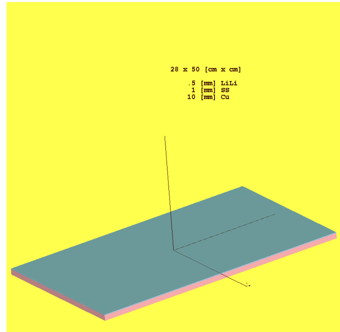
For ST-0, the criterion for readiness of the machine to LiWall regime can be well-defined as:

Demonstration of complete depletion of the plasma discharge by wall pumping, as on T-11M in 1998

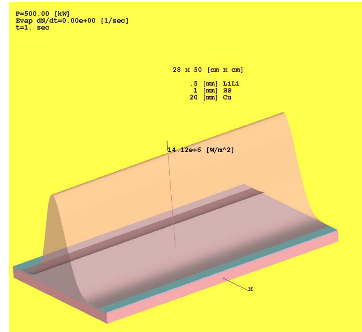
Molten Li is necessary to provide 10000 active monolayers or $\simeq 3\mu k$ of Li.



Li coated plate in low inner divertor



Li/SS/Cu (0.5mm/1mm/10mm) sandwich with a trenched surface

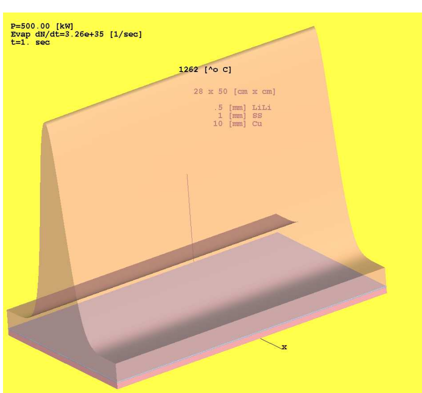


Gaussian (8 cm wide) heat deposition profile

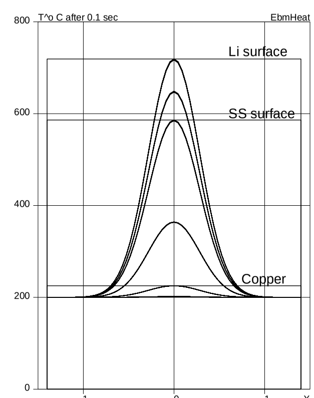
$$S \simeq 0.75\ [m^2], \quad L_{SOL,m} = 2.5, \quad V_{Li} \simeq 0.35\ [L], \quad M_{Li} \simeq 175\ [g],$$
$$\nu_{Pa\cdot sec} = 4.2 \cdot 10^{-4}, \quad I_{ion,MA} = \frac{(0.4-1)\cdot 10^{-3}}{1.6},$$
$$V_{Li,cm/sec} = (2-5)\cdot B_{tor}\frac{h^2_{Li,mm}}{0.01}\frac{0.1}{w_{SOL}}\frac{I_{SoL,MA}}{I_{ion}} \quad (3.1)$$

Li/SS/Cu plate could be real first step toward Li PFC and LiW regime

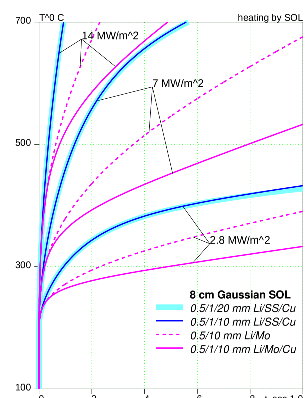
Plate can have different thermal inertia regimes



Surface temperature profile after 0.1 sec



Temperature profile inside the plate



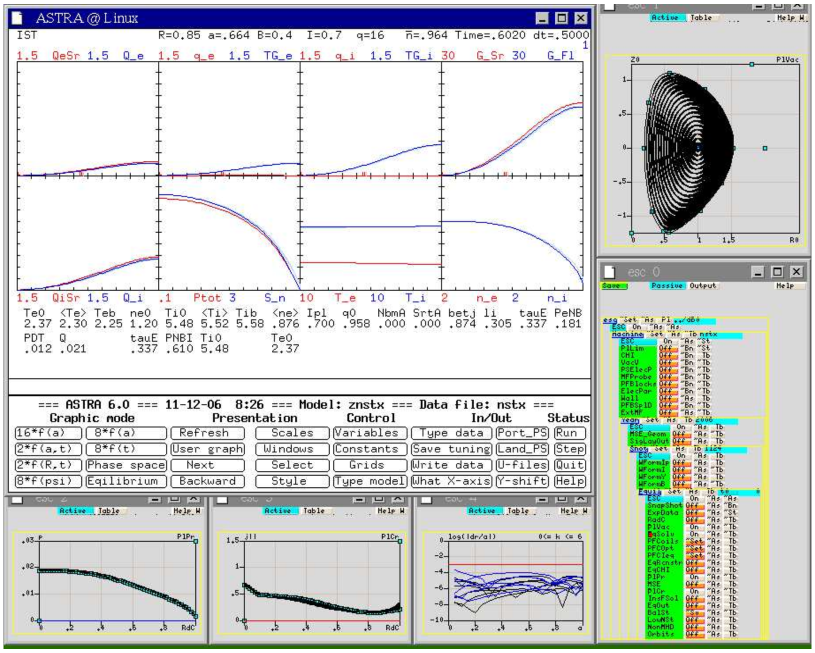
Waveform of the surface temperature

Three cases with 2.5, 1.25, 0.5 MW from the SOL to the plate

Power deposition can be used potentially for maintenance of the Li surface.

SS layer limits the heat transport into the plate body

ASTRA-ESC simulations of ST-0, B=0.4 T, I=0.7 MA, 0.6 MW, 20 keV NBI



Hot-ion mode:

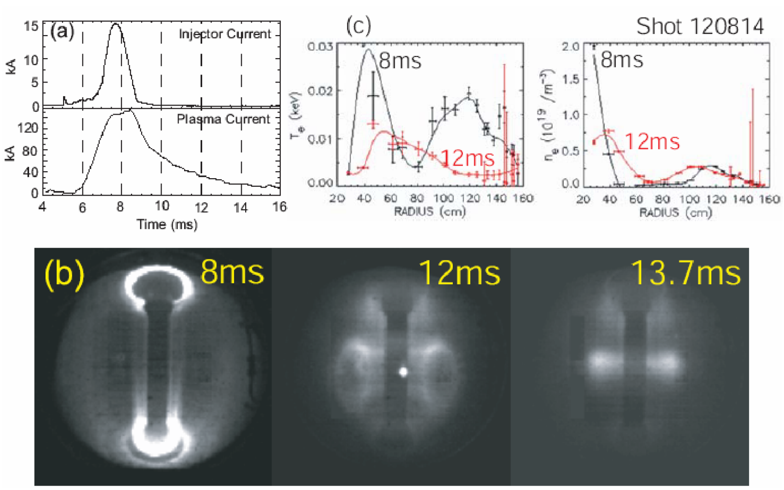
$$T_i = 5.5 \text{ [keV]},$$
$$T_e = 2.5 \text{ [keV]},$$
$$n_e(0) = 0.14 \cdot 10^{20},$$
$$\tau_E = 0.33 \text{ [sec]},$$
$$P_{NBI} = 0.61 \text{ [MW]}$$

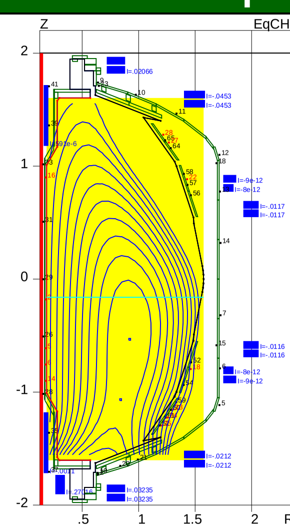
NBI energy should be consistent with the plasma temperature:

$$E_{NBI} = 2.5(T_i + T_e)$$

ST-0 should reach at least 1/3 of τ_E predicted by the Reference Model

LiWF is compatible with both inductive and CHI start-up



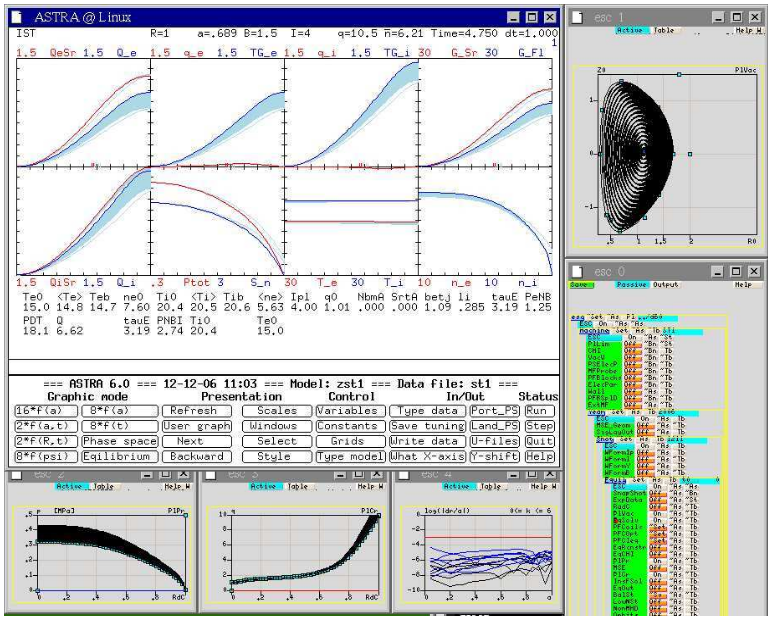


In 2006 CHI startup generated 160 kA current in NSTX

From R.Raman at al., PPPL-4207 (2007)

With Li electrodes, even in the worst case scenario, CHI will create a perfect, transient Li plasma with Z_{eff}=3 (typical for C-wall machines)

ASTRA-ESC simulations of ST-1, B=1.5 T, I=4 MA, 2 MW, 80 keV NBI

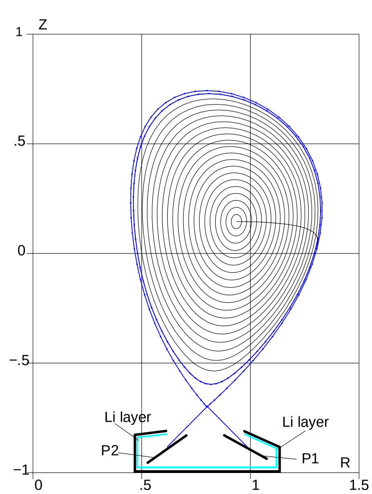


Hot-ion mode:

$\beta = 0.35$,
$T_i = 20$ [keV],
$T_e = 15$ [keV],
$n_e(0) = 0.75 \cdot 10^{20}$,
$\tau_E = 0.34$ [sec],
$P_{NBI} = 2.7$ [MW],
$P_{DT}^{equiv} = 18$,
$Q_{DT}^{equiv} = 6.6$

ST-1 could be the first machine in super-critical regime, $Q_{DT}^{equiv} > 5$

ST-1 should explore all possible divertor options for Li PFC



Sketch of the divertor space with Li inner wall surface

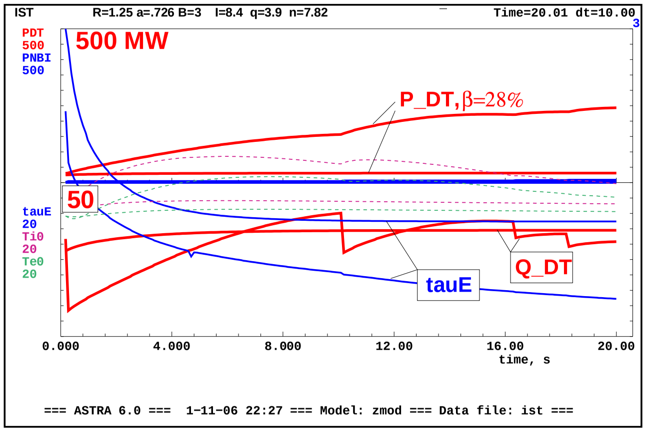
In option V the divertor space is enclosed into a box with the inner walls wetted with liquid Li at low temperature ($<$ 400^o C). The idea is to absorb the D-atoms reflected from the plates.

Advantages with respect to option IV:

1. *Any material can be used as the target surface, while still providing low recycling of D;*
2. *It is not sensitive to the angle between the separatrix and the plates. Hot spot are allowed;*
3. *In case of Li surface, evaporation regime is possible;*
4. *It opens variety of options for suppressing electron emission (if it will be necessary).*

If the design of ST would allow option V, all other options are possible

ASTRA-ESC simulations of IST, B=3 T, I=8.4 MA, 80 keV NBI



$P_{DT}^{equivalent} \simeq 250$ MW,
$\beta = 28\,\%$,
$Q_{DT}^{equivalent} \simeq 40$,
$P_{NBI} < 6$ MW,
$\tau_E = 5 - 16$ sec

The heat load of divertor plates is small

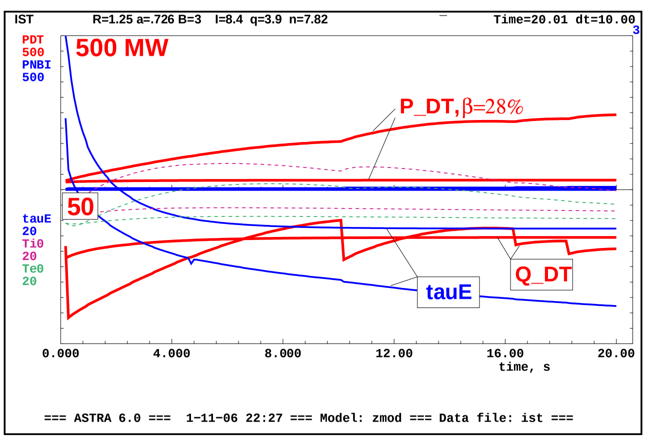
$$P_{NBI} \simeq 6 \text{ MW}$$

The regime of ST-2 (with no fueling by tritium) is identical to RDF

The mission of ST-2 is complete development of the stationary plasma regime for its DT-clone RDF (except extraction of α-particles).

Only LiWF approach allows the development of the full regime for RDF (in Princeton area) with no fueling by tritium

ASTRA-ESC simulations of IST, B=3 T, I=8.4 MA, 80 keV NBI



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The heat load of divertor plates is small

$$P_{NBI} \simeq 6 \text{ MW}$$

Having 30 times smaller volume, RDF can complement ITER with the high fusion power density, neutron flux, and fluence

At $\beta = 40\%$ (0.5 GW) RDF becomes self-sufficient in bootstrap current, free of TEM and, theoretically, capable of DD fusion.

LiWF is compatible with DD fusion and expulsion of energetic p, T, α

This talk is a part of public discussion of one of responses to Orbach/Bodman (DoE) challenge to domestic fusion

In my view, the outlined RDF program has to be considered with equal rights as a competitor to FDF (from GA) and NHTX (from PPPL) and be given the same opportunities for its exposure to the fusion community.

Returning to D.R.: "We should fund projects that reflect revolutionary ideas that, if proven, would make possible the impossible within performance-based guidelines". - So, the problem with funding is not in the Congressmen with extreme views.

Despite some crucial differences in the approaches, the goal is to move fusion forward.

Finally, all of them may merge into a confident next step for magnetic fusion in the US

With its experience in ST and stellarator physics, and no heavy commitment to ITER, PPPL is in unique position for ST-0, ST-1, ST-2 phases of RDF program

Hmmm. Is Leonid really right about Lithium?

NSTX

A new-born fusion thinker, Jillian Maingi

Might be not with 20 % accuracy, but in any case I need a good future and reactor relevant no-T-plasma regimes for my dad $(not\ LZ)$ **in PPPL**